Appendix O
Can-in-Canister Variants

O.1 INTRODUCTION

This appendix presents descriptive information on variants to the vitrification and ceramic immobilization disposition alternatives described as "a can-in-canister approach at Savannah River Site (SRS)" in Table 2.4-1. Based upon comments from the public on the Draft Programmatic Environmental Impact Statement, there is substantial interest in the can-in-canister (CIC) concept for the disposition of surplus plutonium (Pu), and several requests have been made for the Department of Energy (DOE) to consider this concept in its decisionmaking process.

During the initial 12 scoping meetings held across the country from August to October 1994, there was public input on the potential reasonable alternatives and proposed screening process. One of the 37 disposition options under consideration at that time was Option I-2: Borosilicate Glass Immobilization using a Modified Defense Waste Processing Facility (DWPF). This option was eliminated as unreasonable because the DWPF was not designed for criticality control, and therefore would require extensive modifications and refitting of the facility and equipment during its present mission, potentially resulting in increased personnel radiation exposure, along with potential delays and cost escalation for its present mission (DOE 1995m:1-9).

The Department recognized that there could be potential cost savings for the immobilization category of disposition alternatives if the output product from the DWPF (that is, vitrified high-level waste [HLW]) could be used in the disposal of Pu. In response, SRS and the national laboratories developed a CIC concept, which was mentioned in the *Summary Report of the Screening Process* as a variant of Option I-3: Immobilization in Borosilicate Glass (DOE 1995m:4-6). This concept currently consists of two CIC variants that are included in the *Technical Summary Report for Surplus Weapons-Usable Plutonium Disposition* (DOE/MD–0003) dated July 17, 1996.

The CIC concept at SRS is described as an example of a technology variant at an existing facility. However, the Record of Decision (ROD) for this PEIS will only select broader technology strategies for disposition. Site specific decisions and variant-specific decisions will be made pursuant to subsequent *National Environmental Policy Act* (NEPA) review tiered from this initial PEIS.

O.2 CAN-IN-CANISTER CONCEPT AT SAVANNAH RIVER SITE

The CIC concept includes variations to the two Pu disposition alternatives for vitrification and ceramic immobilization. CIC could utilize existing facilities at SRS to house the processes for pit disassembly/conversion, Pu conversion, and vitrification or ceramic immobilization. These existing facilities include the 221-F Facility in F-Area where Pu would be immobilized into a ceramic or glass form and loaded in a can and the DWPF in S-Area where the immobilized Pu would be loaded into a canister with vitrified HLW. For vitrification CIC, Pu would be immobilized in a borosilicate glass matrix in small cans and the cans placed in stainless steel canisters, which are then filled with molten borosilicate glass containing HLW to serve as the radiation barrier. For ceramic CIC, Pu would be immobilized in a ceramic matrix in small cans in lieu of the borosilicate glass, and the cans placed in stainless steel canisters, which are then filled with molten borosilicate glass containing HLW to serve as the radiation barrier. In both cases, the stainless steel canisters would be filled at the DWPF and placed in interim onsite storage at SRS until shipment to a HLW repository is possible.

The CIC concept at SRS could offer the following advantages over the base case immobilization category alternatives:

- Maximize use of existing SRS facilities
- Use vitrified HLW already slated for disposal in a HLW repository
- Provide a simple, yet effective, means to control criticality (that is, small cans of Pu)
- Eliminate unnecessary packaging between front-end operations and immobilization operations
- Require fewer additional canisters in a HLW repository
- Potentially reduce cost and worker radiation exposure

O.3 VITRIFICATION CAN-IN-CANISTER VARIANT

The vitrification CIC variant would process Pu forms to oxide, immobilize Pu oxide in borosilicate glass, and fill individual stainless steel cans. The filled cans would be loaded onto a frame and placed inside an empty stainless steel canister. The canister would have a cylindrical shape with a diameter of 0.6 meters (m) (2 feet [ft]) and length of 3 m (10 ft), and be identical to the stainless steel canister currently used in the DWPF to hold vitrified HLW, with the exception that the canister head would not be welded to the body until after the canister is loaded with cans of Pu glass. The loaded canister would be transferred to the DWPF facility, where molten HLW glass would be poured inside the canister and around the small cans and allowed to harden. The filled canisters would then be decontaminated, welded closed, and stored on-site in the Glass Waste Storage Building until a HLW repository is available for final disposal. A process schematic and material flow diagram for the vitrification CIC process are presented in Figures O.3–1 and O.3–2, respectively.

Facility Description. The vitrification CIC variant could use part of the existing 221-F Canyon building including the Pu Storage Facility and the New Special Recovery facilities, and part of the DWPF including the Vitrification and Service Buildings and the Glass Waste Storage Building. Table O.3–1 lists the location of each process area. Pit disassembly/conversion and other front-end processing, pretreatment operations, and first stage vitrification could be performed in the existing 221-F facility in areas specifically modified to vitrify Pu. The current F-Area facilities are designed and built to handle large quantities of Pu and have systems to maintain criticality control and safeguard systems to maintain accountability and security. The F-Area site layout is shown in Figure O.3–3. The floor area required for the front-end Pu processing and the vitrification functions

Table O.3–1. Locations for Proposed Vitrification Can-in-Canister Process Equipment

Process	Location
Receiving, shipping, storage, sampling	221-F Pu Storage Facility
Pit disassembly, dehydrate/hydride/oxidation	221-F New Special Recovery Facilities
Oralloy decontamination	221-F New Special Recovery Facilities
Special recovery	221-F Canyon 3rd Level
Fuel decladding, halide material processing	221-F Canyon 3rd Level
Feed preparation (dry variants)	221-F Canyon 3rd Level
Oxide lag storage (dry variants)	221-F Canyon 3rd Level
1st stage melter (dry variants)	221-F Canyon 3rd Level
Off-gas treatment (dry variants)	221-F Canyon 3rd Level
Feed preparation (wet variants)	221-F Canyon 3rd Level
Oxide lag storage (wet variants)	221-F Canyon 3rd Level

Table O.3–1. Locations for Proposed Vitrification Can-in-Canister Process Equipment—Continued

Process	Location
1st stage melter (wet variants)	221-F Canyon 2nd Level
Off-gas treatment (wet variants)	221-F Canyon 2nd Level
Can decon (dry variants)	221-F Canyon 3rd Level
Can decon (wet variants)	221-F Canyon 2nd Level
Can weld and test (dry variants)	221-F Canyon 3rd Level
Can weld and test (wet variants)	221-F Canyon 2nd Level
Interim can storage	221-F Canyon 3rd Level
Place in canister	221-F Canyon 1st Level
Weld and test	221-F Canyon 1st Level
Interim canister storage	221-F Canyon 1st Level/DWPF Service Building Interim Vault
Blend tank	DWPF Vitrification Building Hot Cell
2nd stage melter	DWPF Vitrification Building Hot Cell
Canister decontamination	DWPF Vitrification Building Hot Cell
Weld and test	DWPF Vitrification Building Hot Cell
Off-gas treatment	DWPF Vitrification Building Hot Cell
Interim product storage	DWPF Glass Waste Storage Building

Source: LLNL 1996j.

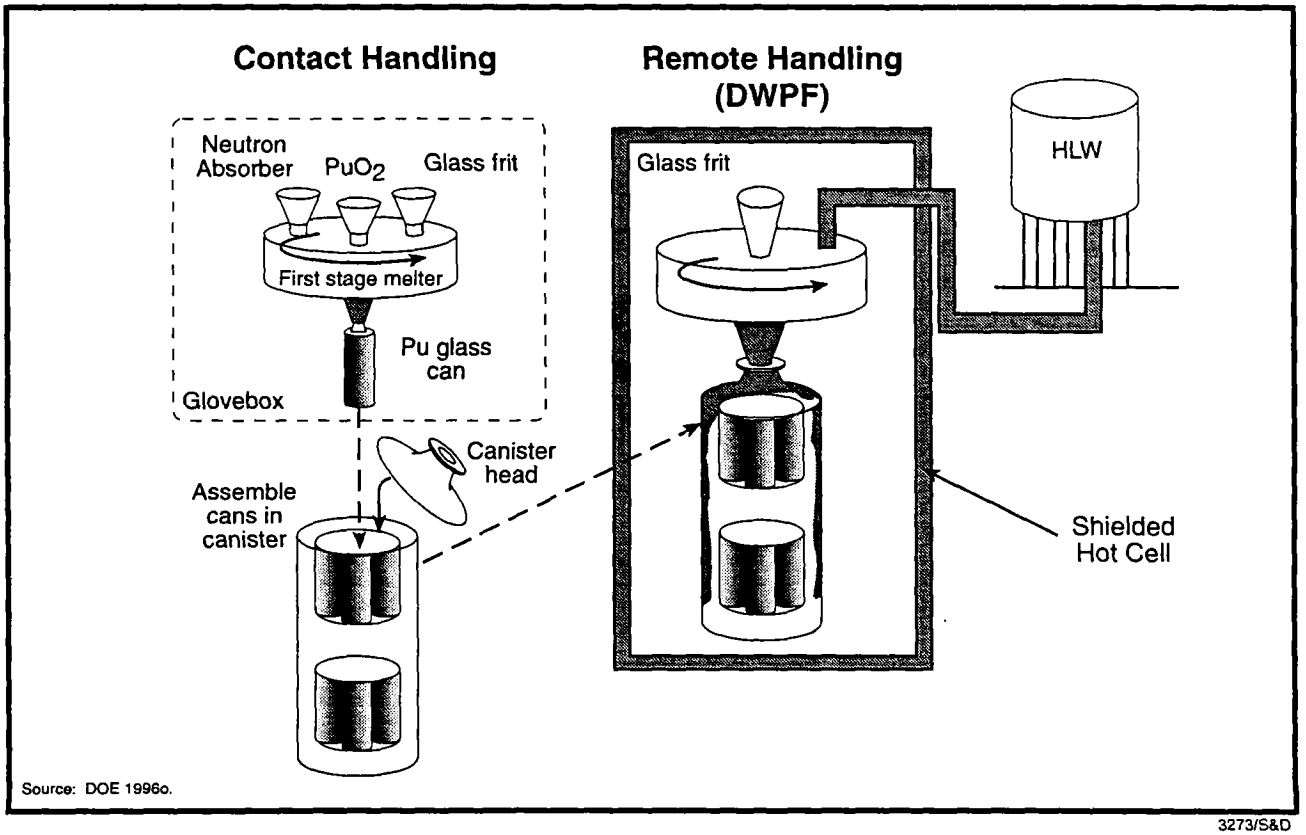

Figure O.3–1. Vitrification Can-in-Canister Variant Process.

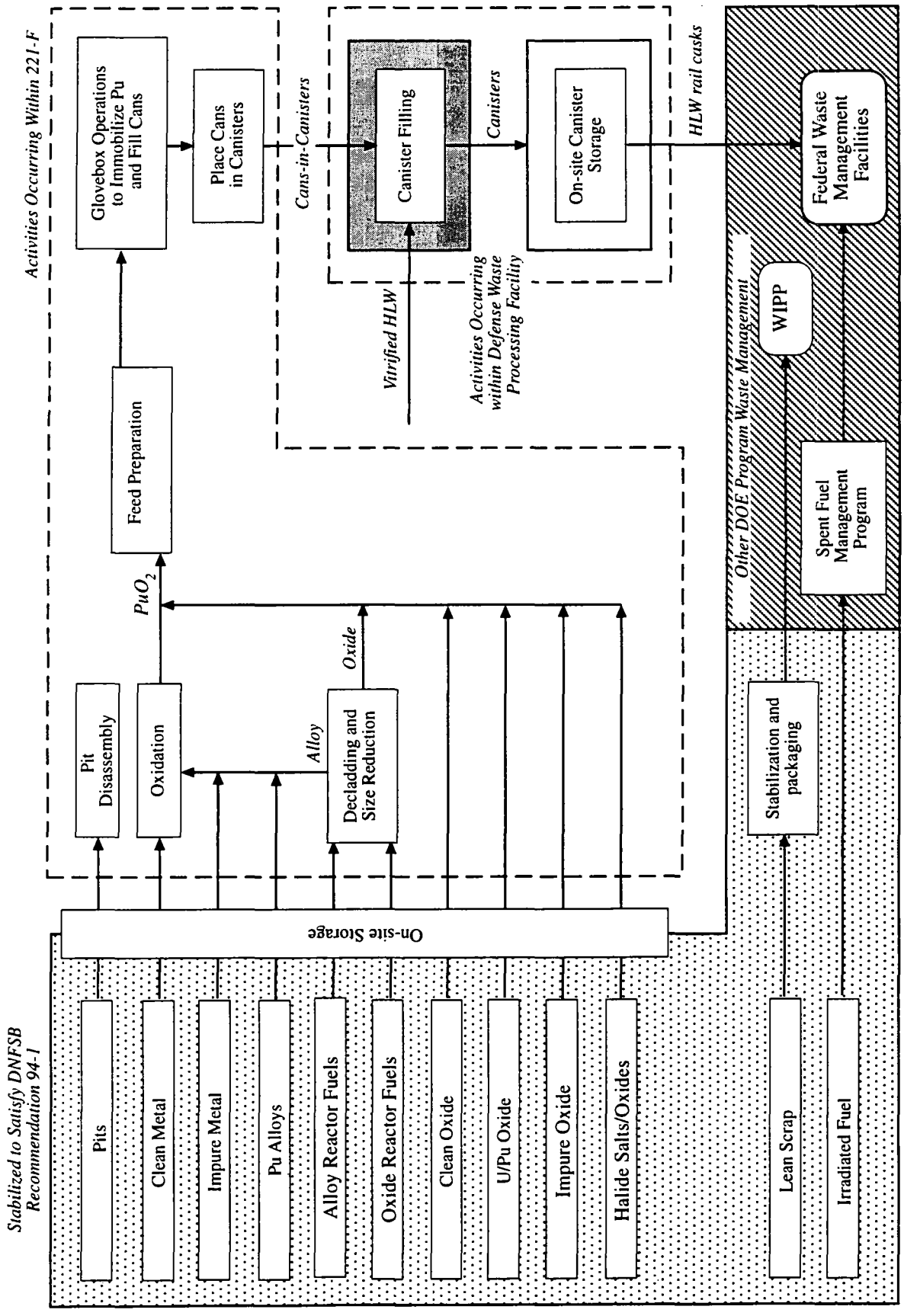

Figure O.3–2. Can-in-Canister Material Flow Diagram for Operations at Savannah River Site.

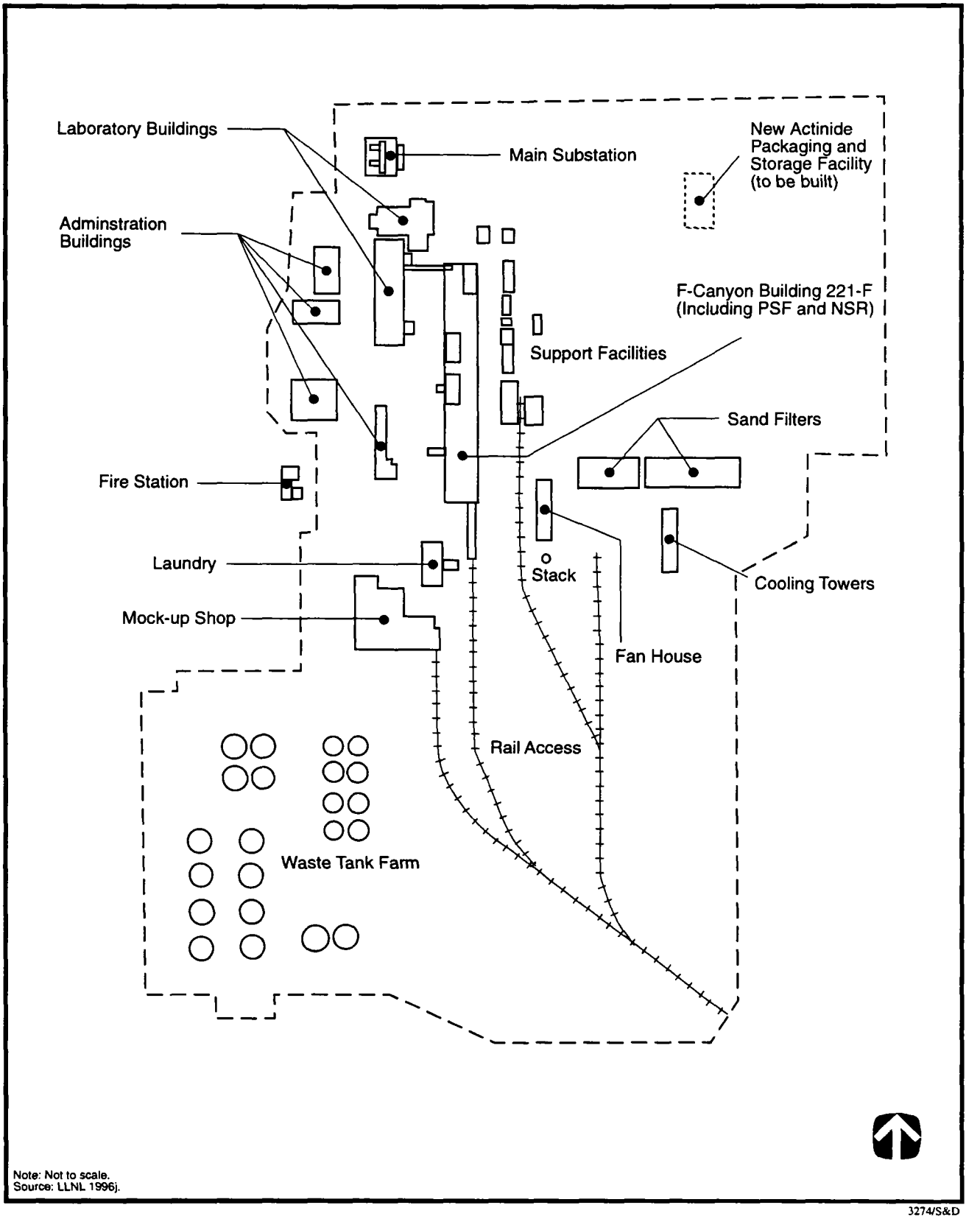

Figure O.3–3. Savannah River Site F-Area Layout Showing Building 221-F.

would be approximately 2,080 square meters (m^2) (22,390 square feet [ft^2]). Total support area (heating, ventilation, and air conditioning, ingress, egress) would be an additional 6,500 m^2 (70,000 ft^2), approximately. The addition of the molten HLW as a radiation barrier in the canister would take place in the DWPF located in the S-Area. The S-Area site layout is shown in Figure O.3–4. The existing DWPF would be upgraded to meet safeguards and security requirements for the handling of stainless steel canisters containing Pu-glass cans during the canister filling operations. Disruptions to current DWPF operations would be minimized since Pu would not be introduced into the DWPF in other than a vitrified, criticality safe manner. Safeguards and security provisions would be upgraded in selected portions of the DWPF buildings.

Facility Operations. In the vitrification CIC variant, the initial step in the immobilization process would be the transportation by safe secure trailer (SST) of Pu feed materials (such as pits, metal, oxides, and unirradiated reactor fuels) from storage site(s) to the receiving facilities in the Pu Storage Facility and New Special Recovery facilities on top of the 221-F Canyon building at SRS. The shipping containers would be unpacked, and accountability measurements conducted. Pu pits would be disassembled and converted to oxide.[1] Other forms would go through the minimum necessary processing to be converted to oxides. The oxide feed materials would then enter into specially modified portions of the 221-F Canyon and undergo first stage immobilization in glove boxes as depicted in Figure O.3–2.

The feeds to this process would consist of glass formers, a neutron absorber, and the Pu oxide which would be combined in a melter to prepare a homogeneous borosilicate glass. This glass would be poured into small stainless steel cans with a Pu concentration of approximately 10 percent by weight. The outside dimensions of the can are a diameter of 13.4 centimeters (cm) (5.3 inches [in]), a length of 47.6 cm (18.7 in), and a shell thickness of 1 cm (0.4 in). After filling, the cans would then be capped, decontaminated, welded, tested, and transferred to lag storage in an onsite secured vault.

When Pu-glass cans are ready for processing, they will be decontaminated and transported to the canister loading area of the F-Canyon. The cans would be placed in a holding rack inside an open DWPF canister (head removed). After loading a canister with approximately 20 Pu-glass cans (5 cans in an array–4 arrays high), the head of the canister would be welded to the body and tested. Each canister would contain approximately 50 kilograms (kg) (110 pounds [lbs]) of Pu. The canisters may be placed in temporary storage in 221-F until they are shipped by rail or truck to the DWPF Service Building. A small storage vault, designed for special nuclear material and sized for about one week supply of canisters, would be provided in or adjacent to the DWPF Service Building to permit interim storage, if necessary, of canisters awaiting processing.

When a canister is ready to be filled, it would be transferred from the DWPF Service Building, via a controlled corridor, to the Melt Cell in the Vitrification Building. Utilizing a melt pour turntable, the HLW-borosilicate glass would be poured into the canister and around the Pu-glass cans. The cans would then become encapsulated in the HLW-glass within the stainless-steel canister. After filling, the canister would be decontaminated and sealed by plug welding and transported to the Glass Waste Storage Facility for interim onsite storage.

The Glass Waste Storage Building, Unit 1, was designed and constructed to hold HLW-glass canisters until a HLW repository is available. Since the capacity of Unit 1 is 2,286 canisters, a Unit 2 building is planned to be constructed as Unit 1 fills with HLW-glass canisters. Since Unit 2 is scheduled to be built in support of the DWPF mission, only upgrades to facility safeguards and security and increased capacity are necessary to accommodate additional canisters resulting from the Pu disposition mission. Alternatively, Unit 1 could be upgraded in the event Unit 2 was determined to be unnecessary. Because the Pu-glass cans displace volume that would normally contain HLW-glass in a separate DWPF operation, additional DWPF canisters would be needed to process all of the HLW in the SRS tank farm and all of the surplus Pu under scope of this PEIS. The number of additional DWPF canisters would be directly proportional to Pu loading in the Pu-glass. The total number of DWPF canisters containing Pu-glass cans would be about 1,000. Assuming 20 cans per canister, the volume of

[1] Based upon the Preferred Alternative, Pu pits would be converted to MOX fuel and would not be immobilized.

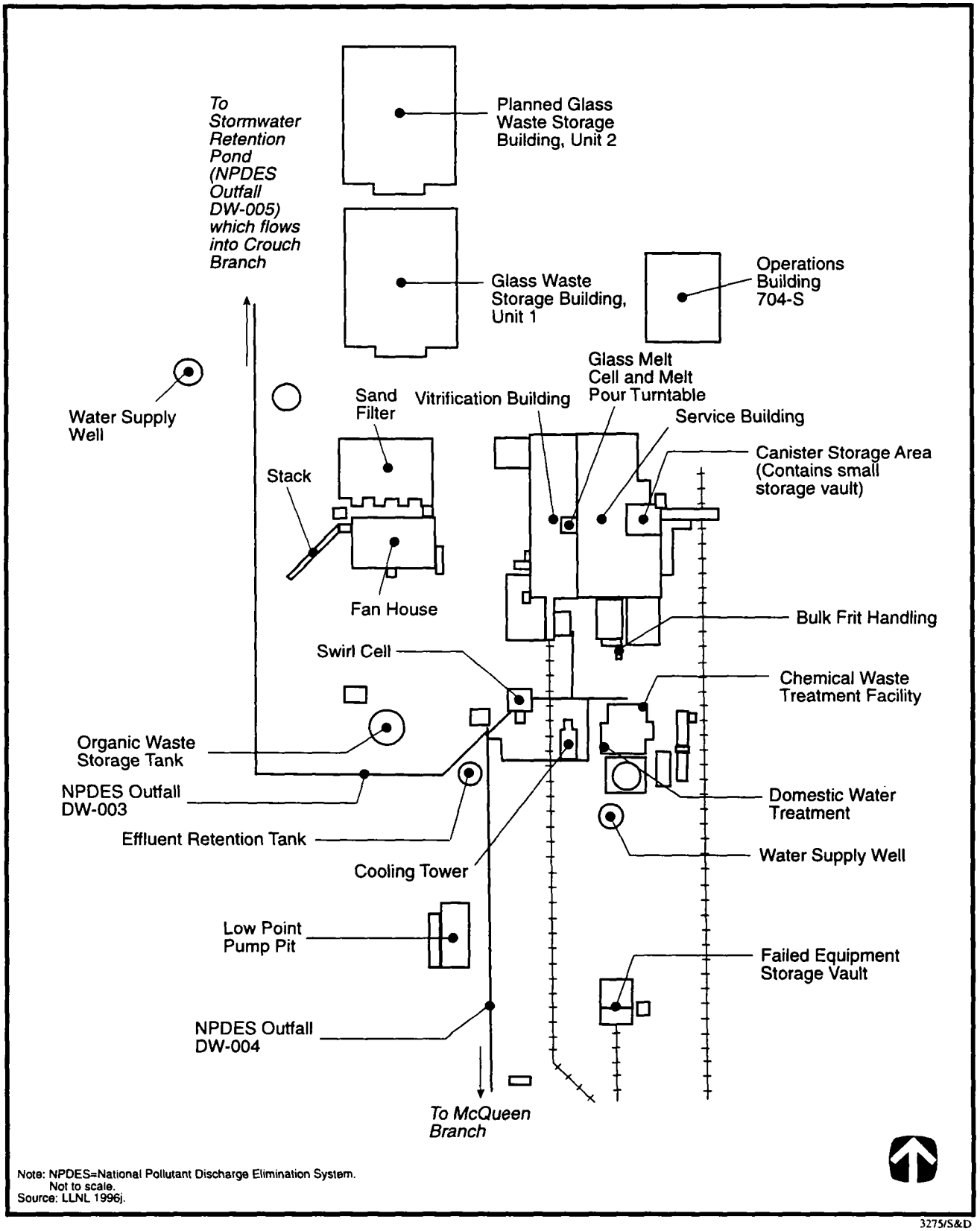

Figure O.3–4. Savannah River Site S-Area Layout Showing Defense Waste Processing Facility.

HLW-glass displaced would be approximately 20 percent, or about 200 additional canisters to those required for the SRS tank farm HLW program.

This variant would process 5,000 kg (11,000 lb) of surplus Pu annually. A normal operating year for the facility would be 200 days with a nominal throughput of 25 kg (55 lb) of Pu per operating day. This operating schedule assumes three shifts per day, 7 days per week. Remote maintenance, accountability, criticality control, and other functions would be performed during the 165 days per year the plant would not be expected to operate.

Construction. The vitrification CIC variant utilizes existing SRS facilities to house the pit disassembly/conversion, Pu conversion, and immobilization operations. For this variant, no major new construction would be required at SRS. However, some of the existing facilities would be modified and upgraded. Facilities in F-Area are designed and built to handle large quantities of Pu and have systems in place to maintain criticality control and safeguards systems to maintain accountability and security. However, the DWPF Vitrification Building, selected portions of the Service Building, and the Glass Waste Storage Building, Unit 1 or 2, would have to be upgraded to meet safeguards and security requirements to support storage and handling of vitrified Pu in accordance with DOE O 470.1 *Safeguards and Security Program.*

Radiological. With the CIC concept, the radiation source used to satisfy the spent fuel standard is the vitrified HLW outside the can. This eliminates the need for introducing radioactive cesium-137 (Cs-137) into the immobilization process, thereby reducing radiation shielding/hot cell requirements and the potential radiation exposures to operating personnel and the public. The dose to workers and the public would be smaller for the CIC variant than the dose to workers and the public for the new facility analyzed in Section 4.3.4.1.9.

Waste Management. Since the Pu conversion and immobilization processes would be similar to those otherwise described for the vitrification alternative, implementation of the vitrification CIC variant would result in similar waste streams. The wastes generated as a result of the operation of this variant would consist of transuranic (TRU), mixed TRU, low-level, mixed low-level, hazardous, nonhazardous (sanitary), and nonhazardous (other) materials. This variant could offer the potential for significant reductions in the quantity of pollutants emitted and wastes generated compared to the construction and operation of the separate disposition facilities as detailed in this PEIS. Significant emphasis would be placed on the minimization of both liquid and solid wastes. Vitrified HLW would be used to surround the cans and fill the canisters; however, no HLW would be generated as a result of the Pu disposition operations. In addition, any criteria pollutants, hazardous air pollutants, and other toxic compounds and gases that may be emitted as a result of the process would be within permit requirements.

Transportation. Fissile material located at various DOE facilities would be transported by SST, in compliance with applicable regulatory requirements, to SRS and placed in onsite temporary storage. Intrasite transport of radiological materials that are not immobilized would be limited to the secure movements within the 221-F Canyon building. Canisters containing the vitrified Pu in cans would be transported from 221-F to the DWPF Service Building via rail or truck. The filled canisters would be stored in the Glass Waste Storage Building until shipment to a HLW repository is possible.

The Department is developing a rail shipping cask for DWPF canisters. This rail shipping cask would hold five DWPF canisters. Based on the use of this transport cask, it is estimated that over the life of the project 40 additional shipments from SRS to a HLW repository would be required.

O.4 CERAMIC CAN-IN-CANISTER VARIANT

The ceramic CIC variant would be similar to the vitrification CIC variant in that both could use existing SRS facilities, produce cans of immobilized Pu, and fill DWPF canisters with Pu and vitrified HLW. The major difference between the two variants is that the Pu inside the stainless steel can would be immobilized in a titanate-based ceramic matrix, rather than a glass matrix. The ceramic product would be formed using a dry

feed, cold press, and sintering (heating) process without including Cs-137 in the ceramic matrix, rather than the wet feed, hot press process with the added Cs-137 radiation barrier as described for the ceramic immobilization alternative in Section 2.4.4.2. The advantages of using the cold press and sintering (heating) process would include increased throughput, simplicity, and proven production experience as used in the MOX fuel industry. Cold pressing would be an option for the ceramic CIC because the volatility of Cs-137 in the sintering process is not an issue with external radiation barrier variants. Using HLW as the radiation barrier, in lieu of the Cs-137 from cesium chloride capsules, would offer the advantages of process simplification and cost reduction, reduction in the potential for worker radiation exposure, and improvements in facility operations and maintenance requirements. If desired, the wet feed and cold press and sintering process could be used in the ceramic CIC variant.

A process schematic for the ceramic CIC process is presented in Figure O.4–1; the material flow diagram for this CIC process is the same as that shown in Figure O.3–2. As in the vitrification CIC variant, many of the feed materials require conversion to oxide form. Such treatment, conversion processing, and oxidation would take place in glove boxes. The resulting Pu oxide product would be fed to the ceramic process where Pu oxide would be blended with ceramic precursors and neutron absorbers. This mixture would be calcined, cold pressed, and sintered to produce densified pellets to be loaded into the small stainless steel cans. The cans of immobilized Pu in ceramic forms would be placed on a frame which would fit inside a DWPF canister and be transferred to the DWPF. These canisters would be filled with vitrified HLW at the DWPF to provide a radiation barrier for the final product.

Facility Description. The ceramic CIC variant could use part of the existing 221-F Canyon building including the Pu Storage Facility and New Special Recovery facilities, and part of the DWPF including the Vitrification and Service Buildings and the Glass Waste Storage Building. Table O.4–1 lists the location of each process area. Pit disassembly/conversion and other front-end processing, treatment operations, and immobilization (feed preparation, calcine and fill, press and package, and can filling) could be performed in the existing 221- F facility in areas specifically modified for Pu ceramic immobilization. The current F-Area facilities are designed and built to handle large quantities of Pu and have systems to maintain criticality control and safeguard systems to maintain accountability and security. The F-Area site layout was previously shown in Figure O.3–3. The floor area required for the front-end Pu processing and immobilization functions would be approximately 2,080 m^2 (22,390 ft^2). Total support area (heating, ventilation, and air conditioning, ingress, egress) would be an additional 6,500 m^2 (70,000 ft^2). The addition of the molten HLW as a radiation barrier in the canister would take place in the DWPF located in the S-Area. The S-Area site layout was previously shown in Figure O.3–4. The existing DWPF would be upgraded to satisfy the safeguards and security requirements for the handling of the stainless steel canisters containing Pu ceramic cans during the canister filling operations. Disruptions to current DWPF operations would be minimized since Pu would not be introduced into the DWPF in other than an immobilized, criticality safe manner. Safeguards and security provisions would be upgraded in selected portions of the DWPF buildings.

Facility Operations. In the ceramic CIC variant, the initial step in the immobilization process would be the transportation by SST, in Department of Transportation shipping containers, of Pu feed materials (such as pits, metal, oxides, and unirradiated reactor fuels) from storage site(s) to receiving facilities in the Pu Storage Facility and New Special Recovery facilities on top of the 221-F Canyon building at SRS. The shipping containers would be unpacked and accountability measurements conducted. Pu pits would be disassembled and converted to oxide.[2] Other forms would go through the minimum necessary processing to be converted to oxides. The oxide feed materials would then enter into specially modified portions of the 221-F Canyon and undergo ceramic immobilization in glove boxes as depicted in Figure O.3–2.

The feeds to this process would consist of ceramic precursors with a neutron absorber, dried titanate ion exchanger, and size-reduced Pu oxide powders which would be dry blended. Dry blending would be conducted

[2] Based upon the Preferred Alternative, Pu pits would be converted to MOX fuel and would not be immobilized.

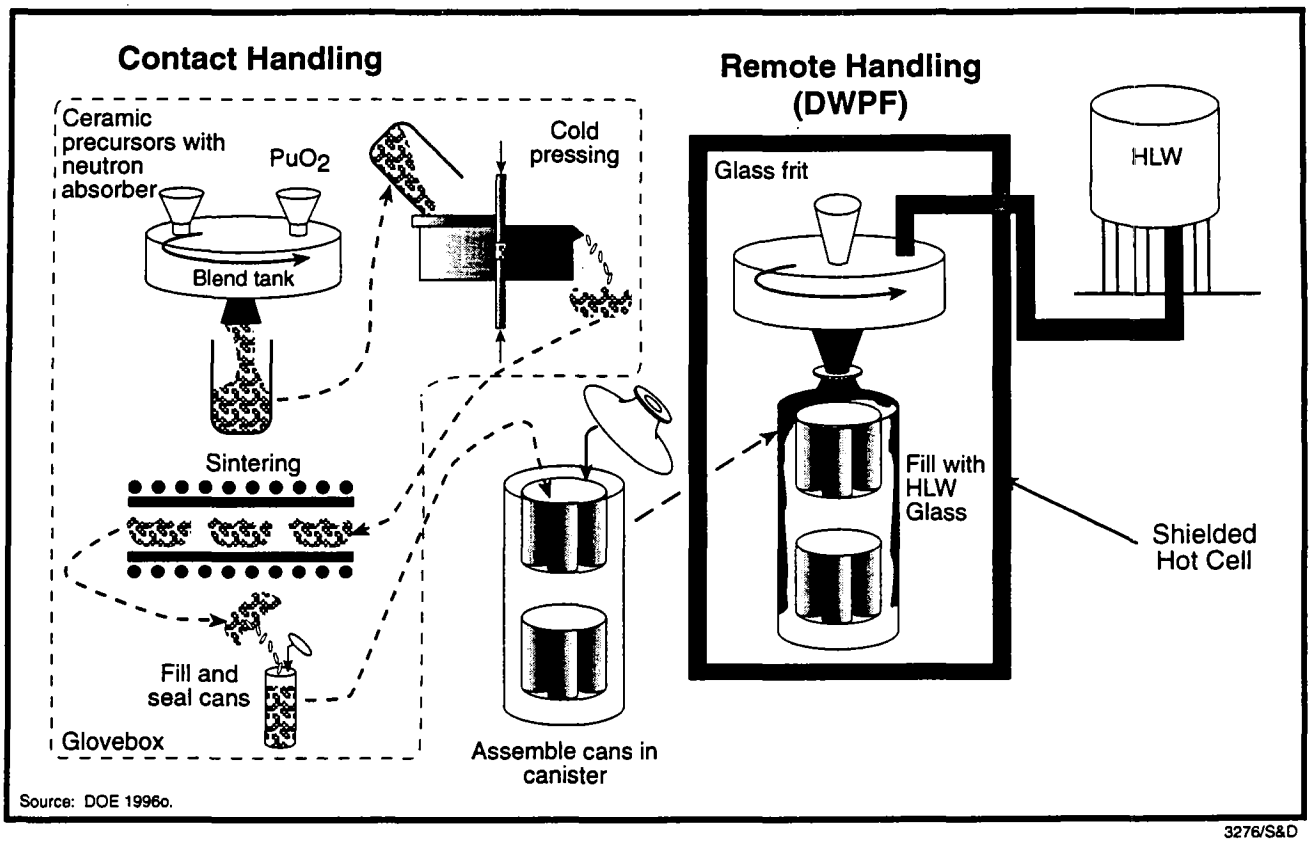

Figure O.4–1. Ceramic Can-in-Canister Variant Process.

Table O.4–1. Locations for Proposed Ceramic Can-in-Canister Process Equipment

Process	Location
Receiving, shipping, storage, sampling	221-F Pu Storage Facility
Pit disassembly, dehydride/hydride/oxidation	221-F New Special Recovery Facilities
Oralloy decontamination	221-F New Special Recovery Facilities
Special recovery	221-F Canyon 3rd Level
Fuel decladding	221-F Canyon 3rd Level
Feed preparation (dry feed)	221-F New Special Recovery Facilities
Oxide lag storage (dry feed)	221-F Canyon 3rd Level
Ceramic press and sinter (dry feed)	221-F Canyon 3rd Level
Off-gas treatment (dry feed)	221-F Canyon 3rd Level
Feed preparation (wet feed variant)	221-F Canyon 3rd Level
Oxide lag storage (wet feed variant)	221-F Canyon 3rd Level
Ceramic press and sinter (wet feed variant)	221-F Canyon 2nd Level
Off-gas treatment (wet feed variant)	221-F Canyon 2nd Level
Can decon (dry)	221-F Canyon 3rd Level
Can decon (wet feed variant)	221-F Canyon 2nd Level
Can weld and test (dry)	221-F Canyon 3rd Level
Can weld and test (wet feed variant)	221-F Canyon 2nd Level
Interim can storage	221-F Canyon 3rd Level
Place in canister	221-F Canyon 1st Level

Table O.4–1. Locations for Proposed Ceramic Can-in-Canister Process Equipment—Continued

Process	Location
Weld and test	221-F Canyon 1st Level
Interim canister storage	221-F Canyon 1st Level/DWPF Service Building Interim Vault
Blend tank	DWPF Vitrification Building Hot Cell
DWPF melter	DWPF Vitrification Building Hot Cell
Canister decontamination	DWPF Vitrification Building Hot Cell
Weld and test	DWPF Vitrification Building Hot Cell
Off-gas treatment	DWPF Vitrification Building Hot Cell
Interim product storage	DWPF Glass Waste Storage Building

Source: LLNL 1996k.

in a standard blending device such as a V-blender. The drying and calcining would be conducted in a rotary calciner that is a rotating tank inside a high temperature furnace. Following the drying and calcining process, the material would undergo milling and granulation to optimize size and morphology for the pressing and sintering operations. The dried and calcined ceramic precursor material loaded with Pu would then be poured into a feeder hopper which would deliver the oxide material into an automated pressing machine. The size of the pellets would be about 7 cm (2.75 in) in diameter by 2.8 cm (1.1 in) high. Pressed pellets would be transferred by a conveyer belt to the sintering oven and heated to 1,200 degrees Centigrade (°C) (2,200 degrees Fahrenheit [°F]) for several hours. After sintering, the pellets would be approximately 6.4 cm (2.5 in) in diameter by 2.5 cm (1 in) high. Any cracked or deformed pellets would be recycled. The Pu-ceramic pellets would then be loaded into 7.6 cm (3 in) diameter by 0.6 m (2 ft) high cans. The Pu loading in the ceramic form would not exceed 12 percent by weight. Cans loaded with the Pu-ceramic would be stored in storage racks in a vault on the third level of the 221-F Canyon building until ready to be placed in the DWPF canister.

When Pu-ceramic cans are ready for processing, they will be decontaminated and transported to the canister loading area of the F-Canyon. The cans would be placed in a holding rack inside an open DWPF canister (head removed). After loading a canister with approximately 20 Pu ceramic cans (5 cans in an array–4 arrays high), the head of the canister would be welded to the body and tested. Each canister would contain approximately 50 kg (110 lb) of Pu. The canisters may be placed in temporary storage in 221-F until shipped by rail or truck to the DWPF Service Building. A small storage vault, designed for special nuclear material and sized for about one week supply of canisters, would be provided in or adjacent to the DWPF Service Building to permit interim storage, if necessary, of canisters awaiting processing.

When a canister is ready to be filled, it would be transferred from the DWPF Service Building via a controlled corridor, to the Melt Cell in the Vitrification Building. Utilizing a melt pour turntable, the HLW-borosilicate glass would be poured into the canister and around the Pu-ceramic cans. The cans would then become encapsulated in the HLW-glass within the stainless-steel canister. After filing, the canister would be decontaminated and sealed by plug welding and transported to the Glass Waste Storage Facility for interim onsite storage.

The Glass Waste Storage Building, Unit 1, was designed and constructed to hold HLW-glass canisters until a HLW repository becomes available. Since the capacity of Unit 1 is 2,286 canisters, a Unit 2 building is planned to be constructed as Unit 1 fills with HLW-glass canisters. Since Unit 2 is scheduled to be built in support of the DWPF mission, only upgrades to facility safeguards and security and increased capacity would be necessary to accommodate the additional canisters resulting from the Pu disposition mission. Alternatively, Unit 1 could be upgraded in the event Unit 2 was determined to be unnecessary. Because the Pu-ceramic cans displace volume that would normally contain HLW-glass in a separate DWPF operation, additional DWPF canisters would be needed to process all of the HLW in the SRS tank farm and all of the surplus Pu. The number of additional

DWPF canisters would be directly proportional to Pu loading in the Pu-ceramic. The total number of DWPF canisters containing Pu-ceramic cans would be expected to be about 1,000. Assuming 20 cans per canister, the volume of HLW-glass displaced would be approximately 20 percent, or about 200 additional canisters to those required for the SRS tank farm HLW program.

This variant would process 5,000 kg (11,000 lb) of surplus Pu annually. A normal operating year for the facility would be 200 days with a nominal throughput of 25 kg (55 lb) of Pu per operating day. This operating schedule assumes 3 shifts per day, 7 days per week. Remote maintenance, accountability, criticality control, and other functions would be performed during the 165 days per year the facility would not be expected to operate.

Construction. The ceramic CIC variant utilizes existing SRS facilities to house the pit disassembly/conversion, Pu conversion, and immobilization operations. Under this variant, no major new construction would be required at SRS. However, some of the existing facilities would be modified and upgraded. Facilities in F-Area are designed and built to handle large quantities of Pu and have systems in place to maintain criticality control and safeguard systems to maintain accountability and security. However, the DWPF Vitrification Building, selected portions of the Service Building, and the Glass Waste Storage Building, Unit 1 or 2, would have to be upgraded to meet the safeguards and security requirements to support storage and handling of immobilized Pu, in accordance with DOE O 470.1, *Safeguards and Security Program*.

Radiological. With the CIC concept, the radiation source used to satisfy the spent fuel standard is the vitrified HLW outside the can. This eliminates the need for introducing radioactive Cs-137 into the immobilization process, thereby reducing radiation shielding/hot cell requirements and the potential radiation exposures to operating personnel and the public. The dose to workers and the public would be smaller for the CIC variant than the dose to workers and the public for the new facility analyzed in Section 4.3.4.2.9.

Waste Management. Since the Pu conversion and immobilization processes would be similar to those otherwise described for the ceramic immobilization alternative, implementation of the ceramic CIC variant would result in similar waste streams. The wastes generated as a result of the operation of this variant would consist of TRU, mixed TRU, low-level, mixed low-level, hazardous, nonhazardous (sanitary), and nonhazardous (other) materials. The ceramic CIC variant could offer the potential for significant reductions in the quantity of pollutants emitted and wastes generated when compared to the construction and operation of the other separate disposition facilities as detailed in this PEIS. Significant emphasis would be placed on the minimization of both liquid and solid wastes. Vitrified HLW would be used to surround the cans and fill the canisters; however, no HLW would be generated as a result of the Pu disposition operations. In addition, any criteria pollutants, hazardous air pollutants, and other toxic compounds and gases that may be emitted as a result of immobilization activities would be within permit requirements.

Transportation. Fissile material located at various DOE facilities would be transported by SST, in compliance with applicable regulatory requirements, to SRS and placed in onsite temporary storage. Intrasite transport of radiological materials that are not immobilized would be limited to the secure movements within the 221-F Canyon. Canisters containing the Pu-ceramic cans would be transported from 221-F to the DWPF Service Building via rail or truck. The filled canisters would be stored in the Glass Waste Storage Building, Unit 2, until shipment to a HLW repository is possible.

The Department is developing a rail shipping cask for DWPF canisters. This rail shipping cask would hold five DWPF canisters. Based on the use of this transport cask, it is estimated that, over the life of the project, 40 additional shipments from SRS to a HLW repository would be required.

O.5 REPOSITORY ACCEPTANCE

An analysis has been completed that examines the feasibility of introducing Pu-loaded glass into a HLW repository, using a CIC concept where the DWPF glass is poured into the space between the Pu glass cans and the DWPF canister (DOE 1996d:8-8). The DWPF glass acts as a radiation barrier to theft and diversion; gadolinium is added to the Pu glass cans as a neutron absorber. The analysis presented represents a case where the CIC concept for Pu disposition is a supplement to the disposition of the defense HLW already planned for the repository. The conclusions presented here for the vitrification variant are expected to apply to the ceramic CIC variant.

Regulatory. Any waste form that is accepted for disposal in a geologic repository must comply with the provisions of the *Nuclear Waste Policy Act* (NWPA), as amended. Under Section 2(12)B of the NWPA, the Nuclear Regulatory Commission (NRC) has the authority to certify this waste as eligible for the NWPA geologic repository. Such NRC action or legislative clarification in authorizing legislation will be necessary before this waste form can be considered for disposal in an NWPA repository. The final disposal of this waste form will have to conform to the licensing provisions of the NRC. Further, it is current DOE policy not to accept into the first HLW repository any wastes that include components regulated as hazardous under the *Resource Conservation and Recovery Act* (RCRA) (DOE 1995a:6). The absence of any RCRA-regulated hazardous materials in the final form would have to be demonstrated prior to acceptance into the HLW repository.

Criticality. The effective neutron multiplication factor (k_{eff}) for the intact glass form, assuming credit for the neutron absorbers during the post-closure period, is calculated to be 0.3 which is well below the 0.95 maximum value of k_{eff} allowed (10 CFR 60).

Thermal. The initial heat release from this CIC waste form is mostly from the HLW glass component. Temperatures peak between 30 and 60 years. By 100 years, the radiolytic heat generated by Pu will exceed the thermal output from the HLW. These predicted temperatures are far lower than, and therefore safely away from, the glass transition temperature of 400°C (750°F). The temperature and thermal output from these canisters are unlikely to materially affect the thermal balance of the repository.

Radiation. A comparison of the radiation exposure emanating from a repository waste package containing DWPF HLW glass canisters versus a package containing Pu cans and HLW glass shows that the radiation dose at the waste package surface is 81 roentgen equivalent man (rem)/hour (hr) for the package containing DWPF glass compared to the 30 rem/hr for one with the Pu cans with HLW glass. The radiation level for the package incorporating Pu CIC is below the threshold value for radiolytic corrosion, so no additional thickness of the copper-nickel alloy waste package outer barrier would be required to reduce the radiation to an acceptable level (100 radiation absorbed dose/hr) to protect the waste package from radiolysis-induced corrosion. However, additional shielding would be required to protect workers. Doses at a distance of 2 m (6.6 ft) from the waste package surface show values of 12.5 rem/hr for the DWPF glass and 4.7 rem/hr for the CIC. For emplacement in the repository, only 7 cm (2.75 in) of lead thickness must be added to the CIC waste package underground transporter to reduce the radiation doses to meet the standard allowable dose of 10 millirem/hr at 2 m (6.6 ft) from lateral outer surfaces (49 CFR 173.441) to ensure worker protection versus 10 cm of lead for repository waste package containing DWPF HLW glass canisters.

Releases. The calculated dose contribution to the accessible environment from all the CIC waste packages would be nearly 100 times lower than the calculated peak dose from a repository that contains only commercial spent fuel and HLW. This is to be expected because the repository release would be dominated by the greater quantity of commercial, uranium-based spent nuclear fuel.

O.6 TESTS AND DEMONSTRATIONS

The department has received comments on the Draft PEIS requesting that "pilot plant" studies of the CIC concept be performed. In this regard, DOE has initiated research, development, and testing of various aspects of the CIC concept for both the vitrification and ceramic immobilization alternatives. For example, in January 1996, a cold (without radionuclides) demonstration of the CIC variant was successfully conducted at the DWPF. Small cans containing a high temperature glass with a Pu surrogate were loaded into two full-size DWPF canisters (one canister contained 8 cans and the other 20 cans) which were subsequently filled with surrogate HLW-glass in the DWPF as part of the cold stratup qualification tests of that facility. Other tests are being conducted to demonstrate techniques that could enhance the nonproliferation properties of the final product. Ceramic waste forms have been under development for HLW for many years; however, the application of this technology to the immobilization of Pu is currently developmental.

The DOE plans to continue research to determine whether glass or ceramic is the preferred form for Pu disposition and to establish the optimum Pu concentration and chemical composition of a waste form that can be readily processed and satisfy nonproliferation concerns and perform well after the emplacement in a geologic repository. In addition, developmental efforts are underway to design acceptable processing equipment and controls and to demonstrate on a pilot scale the integration of the individual processing steps. As part of this effort, the Pacific Northwest National Laboratory, Argonne National Laboratory, Lawrence Livermore National Laboratory, and SRS are each contributing to a cooperative, integrated program of testing and evaluation of forms and processes.

Should DOE select either the vitrification or ceramic immobilization disposition alternative or the CIC variant, these activities will provide the information needed to demonstrate technical viability and practicality of the Preferred Alternative for disposition as well as provide information useful for DOE's tiered NEPA reviews (and RODs) regarding the selection of a specific technology variant, location, and/or facilities.